UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 21, 2017

Mr. Peter P. Sena, III
President and Chief Nuclear Officer
PSEG Nuclear LLC - N09
P.O. Box 236
Hancocks Bridge, NJ 08038

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – STAFF ASSESSMENT OF THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM (CAC NOS. MF5149 AND MF5150; EPID L-2014-LRL-0001)

Dear Mr. Sena:

By letter dated August 11, 2014,[1] as supplemented by letters dated May 28, 2015;[2] March 23, 2016;[3] October 5, 2016;[4] January 13, 2017;[5] and May 3, 2017,[6] PSEG Nuclear LLC (the licensee) submitted a reactor vessel internals (RVI) aging management program (AMP) for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The RVI AMP was submitted to fulfill License Renewal Commitment 7 for Salem, as documented in Appendix A of NUREG-2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station."[7] The RVI AMP is based on "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)."[8]

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the licensee's RVI AMP is provided in the enclosed staff assessment. The NRC staff concludes that the licensee's RVI AMP is acceptable because it is consistent with the Inspection and Evaluation Guidelines of MRP-227-A, and the licensee has adequately addressed the eight specified licensee action items.

The NRC staff's approval of the Salem, Unit Nos. 1 and 2, RVI AMP does not reduce, alter, or otherwise affect current American Society of Mechanical Engineers Code, Section XI, inservice inspection requirements, or any Salem, Unit Nos. 1 and 2, specific licensing requirements related to inservice inspections.

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML14224A667
[2] ADAMS Accession No. ML15148A426
[3] ADAMS Accession No. ML16083A194
[4] ADAMS Accession No. ML16279A092
[5] ADAMS Accession No. ML17013A251
[6] ADAMS Accession No. ML17123A070
[7] ADAMS Accession No. ML11166A135
[8] ADAMS Package Accession No. ML120170453

If you have any questions concerning this matter, please contact the Project Manager, Carleen Parker, at 301-415-1603 or Carleen.Parker@nrc.gov.

Sincerely,

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-272 and 50-311

Enclosure:
Staff Assessment

cc w/Enclosure: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AGING MANAGEMENT PROGRAM OF REACTOR VESSEL INTERNALS

PSEG NUCLEAR LLC

SALEM NUCLEAR GENERATING STATION, UNIT NOS 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated August 11, 2014 (Reference 1), as supplemented by letters dated May 28, 2015 (Reference 2); March 23, 2016 (Reference 3); October 5, 2016 (Reference 4); January 13, 2017 (Reference 5); and May 3, 2017 (Reference 6), PSEG Nuclear LLC (PSEG or the licensee) submitted a reactor vessel internals (RVI) aging management program (AMP) for the Salem Nuclear Generating Station (Salem), Unit Nos. 1 and 2. The RVI AMP is based on "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)" (Reference 7). The RVI AMP was submitted to fulfill License Renewal Commitment 7 for Salem, as documented in Appendix A of NUREG-2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station" (Reference 8). The AMPs include Inspection and Evaluation (I&E) Guidelines for the RVI components at Salem, Unit Nos. 1 and 2, during the period of extended operation (PEO).

Attachment 1 of the August 11, 2014, submittal contains the Salem, Unit No. 1, RVI AMP.
Attachment 2 of the August 11, 2014, submittal contains the Salem, Unit No. 2, RVI AMP.

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for renewal of operating licenses for nuclear power plants," addresses the requirements for plant license renewal process. The regulation at 10 CFR 54.21, "Contents of application – technical information," requires that each application for license renewal contain an integrated plant assessment and an evaluation of time-limited aging analyses. The plant-specific integrated plant assessment shall identify and list those structures and components subject to an aging management review and demonstrate that the effects of aging (e.g., cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis during the PEO as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22, "Contents of application – technical specifications," requires a license renewal application to include any technical specification changes or additions necessary to manage the effects of aging during the PEO as part of the license renewal application.

Enclosure

Structures and components subject to an AMP shall encompass those structures and components that are referred to as "passive" and "long-lived." Passive structures and components perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties. Long-lived structures and components are not subject to replacement based on a qualified life or specified time period. The scope of components considered for inspection under MRP-227-A includes core support structures typically denoted as Examination Category B-N-3 by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, and those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include consumable components such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMP, as defined by the criteria set forth in 10 CFR 54.21(a)(1).

2.2 Licensee Renewal Commitment

The NRC issued NUREG-2101 in June 2011. Appendix A of NUREG-2101 included license renewal Commitment 7 related to the Salem, Unit Nos. 1 and 2, RVI AMPs. Specifically, the licensee committed to the following activities: (1) participate in the industry programs for investigating and managing aging effects on reactor internals, (2) evaluate and implement the results of the industry programs as applicable to the reactor internals, and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval. PSEG developed the Salem, Unit Nos. 1 and 2, RVI AMPs based on MRP-227-A.

On January 12, 2009, Electric Power Research Institute (EPRI) submitted, for NRC staff review and approval, MRP-227, Revision 0 (Reference 9), which was intended as guidance for applicants in developing their plant-specific AMPs for RVI components. MRP-227 contains a discussion of the technical basis for the development of plant-specific AMPs for RVI components in pressurized-water reactors (PWRs) vessels and also provides I&E Guidelines for PWR applicants to use in their plant-specific AMPs. The NRC staff issued Revision 1 to its Final Safety Evaluation (SE) for MRP-227 on December 16, 2011 (Reference 10), with seven topical report (TR) conditions and eight applicant/licensee action items. The TR conditions were specified to ensure that certain information was revised generically in the approved version of MRP-227-A and the applicant/licensee action items addressed plant-specific issues that could not be resolved generically in the December 16, 2011, SE. On January 9, 2012, EPRI submitted the NRC-approved version of the TR designated MRP-227-A. To fulfill license renewal Commitment 7 PSEG needs to address the plant-specific issues specified in the eight applicant/licensee action items.

3.0 TECHNICAL EVALUATION

The NRC staff assessment of the Salem, Unit Nos. 1 and 2, RVI AMPs focused on determining whether the licensee adequately incorporated the I&E Guidelines recommended in MRP-227-A and the licensee's resolutions to the applicant/licensee action items and TR conditions. Specifically, the NRC staff assessment focused on the following: (1) the licensee's implementation of the MRP-227-A I&E Guidelines for RVI components in the primary, expansion, and existing categories, as well as the appropriate acceptance criteria; (2) operating experience of RVI component degradation at Salem, Unit Nos. 1 and 2; (3) the licensee's AMP for RVI components under the current ASME Code, Section XI, inservice inspection (ISI) program; and (4) the licensee's resolutions to the eight applicant/licensee action items.

The NRC staff notes that the seven TR conditions are generic conditions imposed on the approval of MRP-227-A, not conditions for individual licensees. The NRC staff reviewed Table 6-1, "Topical Report Condition Compliance to SE on MRP-227," in the Salem RVI AMPs and confirmed that the Salem RVI AMPs are consistent with the seven TR condition resolutions in MRP-227-A.

The following sections provide details of the NRC staff assessment of the four assessment items listed above. By letters dated March 31, 2015 (Reference 11), and July 7, 2016 (Reference 12), the NRC staff issued requests for additional information (RAIs) to support this assessment.

3.1 Assessment Area 1 – MRP-227-A I&E Guidelines for RVI Components in the Primary, Expansion, and Existing Categories and Acceptance Criteria

In Attachments 1 and 2 of its August 11, 2014, submittal, the licensee implemented the MRP-227-A I&E Guidelines in the primary, expansion, and existing categories in the following tables:

- Table C-1, "MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals";
- Table C-2, "MRP-227-A Expansion Inspection and Monitoring Recommendations for Westinghouse-Designed Internals"; and
- Table C-3, "MRP-227-A Existing Inspection and Aging Management Credited in Recommendations for Westinghouse-Designed Internals."

The NRC staff reviewed these tables and determined that they are consistent with the following tables from MRP-227-A and are, therefore, acceptable:

- Table 4-3, "Westinghouse plants Primary components";
- Table 4-6, "Westinghouse plants Expansion component"; and
- Table 4-9, "Westinghouse plants Existing Programs components."

Additionally, the NRC staff reviewed the acceptance criteria in Table C-4, "MRP-227-A Acceptance Criteria and Expansion Criteria Recommendations for Westinghouse-Designed Internals," of Attachments 1 and 2 of the licensee's submittal and determined that the acceptance criteria are consistent with Table 5-3, "Westinghouse plants examination acceptance and expansion criteria," of MRP-227-A, and are, therefore, acceptable.

Based on this review, the NRC staff has determined that the licensee adequately implemented the MRP-227-A I&E Guidelines into the Salem, Unit Nos. 1 and 2, RVI AMPs.

3.2 Assessment Area 2 – Operating Experience of RVI Component Degradation

To address MRP-227-A TR Condition 7, MRP-227-A, Appendix A, "Reactor Internals Operational Experience," was updated to include the operating experience related to the aging degradation of the RVI components in the PWR fleet. The update references Section XI.M16A of NUREG-1801, "Generic Aging Lessons Learned (GALL) Report – Final Report," Revision 2 (Reference 13). MRP-227-A, Appendix A, addresses operating experience for some RVI components that are susceptible to various aging degradation mechanisms. These RVI components are discussed in the subsections that follow. Additionally, the GALL report states that licensees are expected to review subsequent operating experience and evaluate its impact on their AMPs.

3.2.1 Additional RVI Components Susceptible to Degradation

By letter dated March 31, 2015, the staff requested the licensee provide information regarding additional components that are susceptible to aging degradation mechanisms not listed in MRP-227-A and MRP-191, "Screening, Categorization and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design" (Reference 14). MRP-191 is the technical basis for the MRP-227-A categorization of Westinghouse RVI components. By letter dated May 28, 2015, the licensee responded that in support of its resolution to applicant/licensee action item 2 (discussed in Section 3.4.2 of this assessment), the Salem, Unit Nos. 1 and 2, RVI components were reviewed and compared with the materials listed in MRP-191. The licensee concluded from this review that there are no additional RVI components manufactured from the materials listed in the staff's request. Based on this response and the licensee's resolution of applicant/licensee action item 2, the NRC staff concludes that there are no additional RVI components needed for aging management during the PEO.

3.2.2 Operating Experience on Pressurized-Water Stress Corrosion Cracking in Alloy X-750

In its submittal, PSEG states that Salem, Unit Nos. 1 and 2, Alloy X-750 clevis insert bolts are inspected for wear via visual testing (VT-3) under the existing ASME Code, Section XI, ISI program. MRP-227-A, Appendix A, discusses one Westinghouse plant that reported failures of Alloy X-750 clevis insert bolts that was likely caused by pressurized-water stress corrosion cracking (PWSCC). The NRC staff notes that Alloy X-750 components that receive high temperature heat (HTH) treatment offer better resistance to PWSCC than those that receive other age-hardened heat treatment processes. As such, Salem, Units 1 and 2, Alloy X-750 clevis insert bolts could be susceptible to PWSCC if they did not receive HTH and, therefore, the existing ASME Code, Section XI, ISI VT-3 inspection would not be adequate to detect cracking in the bolts.

By letter dated March 31, 2015, the NRC staff requested the licensee provide information related to the heat treatment process used for Alloy X-750 clevis insert bolts at Salem. By letter dated May 28, 2015, the licensee responded that, after reviewing drawings and specifications for the Alloy X-750 clevis insert bolts at Salem, Unit Nos. 1 and 2, the bolts did not receive HTH treatment. In its response, the licensee stated that the clevis insert bolts are included under the existing ASME Code, Section XI, ISI program at Salem, Unit Nos. 1 and 2, and determined that the ASME Code, Section XI, ISI program was adequate to monitor the bolts through the PEO. The licensee stated that it has enhanced the ASME Code, Section XI, ISI requirements with guidance provided in Westinghouse Technical Bulletin, TB-14-5, Revision 0, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation" (Reference 15). The Westinghouse technical bulletin did not change the type or timing of inspections but gave recommendations for the scope and focus of the inspections to detect known indications of failure. PSEG stated that it performed the last ASME Code, Section XI, inspections for the clevis insert bolts in spring 2001 and spring 2002 for Salem, Unit Nos. 1 and 2, respectively. During these inspections, the licensee reported no relevant indications. PSEG stated that it will continue to inspect the clevis insert bolts with the ASME Code, Section XI, ISI program enhanced with the guidance from the Westinghouse technical bulletin.

The NRC staff reviewed the information provided by PSEG and the information in the Westinghouse technical bulletin. The Westinghouse technical bulletin states that Westinghouse performed detailed structural evaluation of the as-found condition of the clevis insert bolts of the one plant that reported failures. The structural evaluation showed that failure of the bolts would not result in a loss of safety function of the surrounding RVI. The NRC staff notes that the function of the clevis inserts (which are secured in the vessel by the clevis insert bolts) are for alignment of the lower RVI assembly and that they do not support the lower RVI assembly weight. When the lower RVI assembly is assembled in the vessel, the clevis inserts are tightly held in place. This means that if the clevis insert bolts fail, the clevis inserts would remain in place and the lower RVI assembly would remain aligned. In addition, there are significant redundancies in the clevis insert assembly (for example there are several bolts that secure one clevis insert) that would prevent the clevis insert assembly from being nonfunctional. Therefore, the NRC staff finds Westinghouse's conclusion that failed bolts would not lead to a loss of safety function of the surrounding RVI reasonable. The NRC staff determined that the structural evaluation discussed in the Westinghouse technical bulletin adequately addresses the concern of PWSCC susceptibility of the Alloy X-750 clevis insert bolts at Salem, Unit Nos. 1 and 2.

As discussed above, the NRC staff determined there would be no loss of safety function of the surrounding RVI even if the Alloy X-750 clevis insert bolts failed due to PWSCC. Therefore, the NRC staff concludes that the VT-3 inspection for wear in the existing ASME Code, Section XI, ISI program for the Salem, Unit Nos. 1 and 2, Alloy X-750 clevis insert bolts is adequate to manage the aging degradation of the bolts.

3.2.3 Inspection of Wear of Control Rod Guide Tube Cards

Appendix A of MRP-227-A includes a generic discussion of wear of the Control Rod Guide Tube (CRGT) cards. Salem, Unit Nos. 1 and 2, RVI AMPs, Table C-1, "MRP-227-A Primary Inspection and Monitoring Recommendations for Westinghouse-Designed Internals," indicates that visual testing (VT-3) for general degradation will be used for the CRGT cards. Table C-1 also indicates the examination schedule recommended in proprietary report WCAP-17451-P, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," will be used. The NRC staff's SE of WCAP-17096 (Reference 16), which contains acceptance criteria for MRP-227-A primary and expansion components, found the examination methodology for CRGT cards acceptable. WCAP-17451-P takes into account operational experience and analyses that were not available during the development of MRP-227-A and includes a more comprehensive inspection coverage for the CRGT cards than the coverage specified in MRP-227-A.

The NRC staff finds that the licensee's use of WCAP-17451-P adequately addresses operating experience of wear in CRGT cards and, therefore, provides reasonable assurance that the Salem, Unit Nos. 1 and 2, CRGT cards will be adequately managed for wear during the PEO.

3.2.4 Salem, Unit No. 1, Baffle-to-Former Bolts Degradation

Salem, Unit No. 1, baffle-to-former bolts degradation was reported as Event Number 51902 in accordance with the reporting requirements of 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors." In response the NRC staff requested the licensee to discuss the reinspection schedule, the justification of the reinspection frequency, the reinspection method, and the reinspection coverage. By letter dated October 5, 2016, the licensee responded that it completed ultrasonic examination volumetric testing (UT) of the full population of the Salem, Unit No. 1, accessible baffle-to-former bolts during refueling outage 24 in spring 2016. This UT inspection was consistent with the recommendations in EPRI letter MRP 2016-022, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL 16-01," (Reference 17) that all Tier 1a plants, which includes Salem, Unit No. 1, perform UT inspection of the full population of baffle-to-former bolts. During refueling outage 24, the licensee replaced 189 baffle-to-former bolts with new bolts as a result of the UT inspection. The licensee also stated that it will continue to evaluate operating experience from other plants performing baffle-to-former bolt inspections and recommendations of the industry focus group on baffle-to-former bolts, and will continue to implement industry guidance. These efforts support the considerations on reinspection frequency, inspection method, and the inspection coverage for baffle-to-former bolts.

By letter dated July 7, 2016, the NRC staff also requested the licensee to discuss the potential impact of the degraded baffle-to-former bolts at Salem, Unit No. 1, on the MRP-227-A inspection program for the Salem, Unit No. 2, baffle-to-former bolts. By letter dated October 5, 2016, the licensee stated it is planning to complete UT inspection of the full population of the Salem, Unit No. 2, accessible baffle-to-former bolts during refueling outage 22 in spring 2017. This planned UT inspection is consistent with the recommendations in EPRI letter MRP 2016-022 that all Tier 1a plants, which includes Salem, Unit No. 2, perform UT inspection of the full population of baffle-former bolts at the next scheduled refueling outage.

The licensee is actively participating in evaluating recent operating experience with baffle-to-former bolt degradation and assessing its impact on the Salem, Unit Nos. 1 and 2, RVI AMPs. The licensee has replaced the degraded Salem, Unit No. 1, baffle-former bolts, and the licensee's proactive efforts regarding baffle-to-former bolt degradation provide reasonable assurance that the licensee will adequately manage the aging of the Salem, Unit Nos. 1 and 2, baffle-to-former bolts during the PEO.

3.2.5 Conclusion of Section 3.2

Based on the discussions in Sections 3.2.1 through 3.2.4 of this assessment, the NRC staff determined that the licensee has adequately considered and addressed operating experience in the Salem, Unit Nos. 1 and 2, RVI AMPs.

3.3 Assessment Area 3 – Salem's AMPs for RVI Components under the ASME Code, Section XI, ISI Program

By letter dated March 31, 2015, the NRC staff requested the licensee to provide a list of RVI components inspected thus far under the ASME Code, Section XI, ISI program and the inspection results. This list included any RVI component categorized under the "Existing" inspection category of the MRP-227-A report. By letter dated May 28, 2015, the licensee listed the RVI components in the "Existing" inspection category, except the flux thimble tubes because they are part of the ASME Code, Section XI, ISI program. PSEG stated that the flux thimble tubes of Salem, Unit Nos. 1 and 2, will be inspected under the Flux Thimble Tube Inspection Program described in the Salem, Unit Nos. 1 and 2, RVI AMPs. The licensee stated that the Flux Thimble Tube Inspection Program, a program the NRC staff concluded was consistent with the GALL Report (Reference 13) in NUREG-2101, will be implemented prior to the PEO.

In addition, the licensee stated that a review of all inspections completed to date for the RVI components in the "Existing" category of MRP-227-A identified no relevant indications. Each of these items has received VT-3 inspection under the ASME Code, Section XI, ISI program. For all other RVI components that have been examined thus far, the licensee stated that the examination results are contained in plant records

Based on the licensee's responses, the NRC staff determined that the licensee is adequately implementing the ASME Code, Section XI, ISI program for the Salem, Unit Nos. 1 and 2, RVI components under the "Existing" category during the PEO.

3.4 Assessment Area 4 - The Eight Applicable Applicant/Licensee Action Items

3.4.1 Assessment of Resolution to Applicant/Licensee Action Item 1

MRP-227-A, Section 4.2.1, "Applicability of Failure Modes, Effects, and Criticality Analyses (FMECA) and Functionality Analysis Assumptions," states:

> As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE [Combustion Engineering], or B&W [Babcock and Wilcox]) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. **This is Applicant/Licensee Action Item 1**.

The purpose of applicant/licensee action item 1 is to determine the applicability of the I&E Guidelines of MRP-227-A to the specific plant, in this case, to Salem, Unit Nos. 1 and 2 EPRI and the MRP have developed guidelines for determining the applicability of MRP-227-A in letter MRP 2013-025 (Reference 18). The NRC staff reviewed MRP 2013-025 and its technical basis contained in proprietary WCAP-17780-P. The NRC staff concluded in its evaluation of WCAP-17780-P (Reference 19) that if a licensee meets the recommendations in MRP 2013-025, the licensee will have demonstrated reasonable assurance that the I&E guidelines of MRP-227-A will be applicable to the specific plant.

MRP 2013-025 lists two questions that all Westinghouse-designed plants must address to resolve applicant/licensee action item 1. The two questions, applicable to Salem, Unit Nos. 1 and 2, are summarized below:

1. Do the Salem RVI components have non-weld or non-bolting* austenitic stainless steel components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 kilopounds per square inch (ksi)?

*Written as "bolting" in MRP 2013-025. The correct term "non-bolting" is made here.

2. Have Salem, Unit Nos. 1 and 2, utilized atypical fuel design or fuel management, including power changes/uprates, that could render the assumptions of MRP-227-A regarding core loading/core design non-representative for the two units?

By letter dated March 23, 2016, PSEG provided a response to these questions. The NRC staff summarizes and assesses the licensee's responses below.

Response to Question 1

The licensee evaluated the potential for greater than 20 percent cold work in the Salem, Unit Nos. 1 and 2, RVI components following the MRP 2013-025, Appendix A, Option 1, guidance. Specifically, the licensee binned the Salem, Unit Nos. 1 and 2, RVI components into five material categories and based the potential for cold work on the material specifications of the ASME Code or the American Society for Testing and Materials. Categories 1, 2, and 3 are non-cold worked components and include cast austenitic stainless steel (CASS), hot-formed austenitic stainless steel, and annealed austenitic stainless steel. Category 4 is austenitic stainless steel fasteners, such as bolting, which are already assumed to have potentially greater than 20 percent cold work in the MRP-191 categorization and are, therefore, already in the proper MRP-227-A component category. Category 5 is cold-formed austenitic stainless steel. The licensee stated that material fabrication processes in Category 5 components prevented the introduction of severe cold work (i.e., greater than 20 percent cold work). The licensee stated that where multiple options existed for a component or assembly (i.e., the component or assembly falls into one or more categories), the option that had the potential for a greater amount of cold work was assumed.

The Pressurized Water Reactors Owner's Group (PWROG) developed report PWROG-15105-NP (Reference 20) to address cold work in PWR RVI components. The NRC staff issued a summary assessment dated April 21, 2017 (Reference 21), of this report and stated that no non-fastener RVI components were subject to cold work greater than 20 percent in Westinghouse PWR units, and that these components, therefore, are less susceptible to PWSCC.

The NRC staff finds this response to be consistent with the guidance in MRP 2013-025 and PWROG-15105-NP, and that the licensee adequately demonstrated that Salem, Unit Nos. 1 and 2, have no RVI components with greater than 20 percent cold work. Thus, the NRC staff determined that the licensee adequately demonstrated that MRP-227-A is applicable to both Salem units with respect to cold work concerns in RVI components.

Response to Question 2

The licensee evaluated the assumptions for Salem, Unit Nos. 1 and 2, regarding core loading/core design, neutron fluence, and heat generation rates following the guidance presented as Option 1(A) in Appendix B of MRP 2013-025. Specifically, the licensee stated that Salem, Unit Nos. 1 and 2, have not used atypical fuel designs or fuel management that could make the assumptions of MRP-227-A non-representative respect to core loading/core design and power changes/uprates.

The NRC staff reviewed the licensee's response and determined that the licensee met the MRP 2013-025 guidelines (with one exception discussed below) recommended for these reactor core regions of Salem, Unit Nos. 1 and 2: the outer radius of the reactor core, above the reactor core, and below the reactor core. The MRP 2013-025 guidelines included limiting threshold values for three parameters in Westinghouse units related to core geometry and core loading, which are listed below:

- Active fuel to upper core plate distance > 12.2 inches
- Average core power density < 124 Watts/cm^3
- Heat generation figure of merit, F ≤ 68 Watts/cm^3

For Salem, Unit Nos. 1 and 2, in the outer radius of reactor core, the average core power density is less than 124 Watts/cm^3, and the heat generation figure of merit is less than 68 Watts/cm^3. In the region above the reactor core, the average core power density is less than 124 Watts/cm^3 for both units, and for Salem, Unit No. 2, the active fuel to upper core plate distance is greater than 12.2 inches. In the region below the reactor core, both units met the general applicability assumptions in Section 2.4, "Guidelines Applicability," of MRP-227-A and, therefore, as recommended in MRP 2013-025, no further evaluation is necessary. Additionally, the licensee stated that Salem, Unit Nos. 1 and 2, have implemented a low leakage core loading pattern and have no plans to return to an out-in core loading pattern.

The one exception in which the MRP 2013-025 guideline is not met is that for Salem, Unit No. 1, the active fuel to upper core plate distance was less than 12.2 inches. The licensee stated that this condition existed for an operating period of more than 2 calendar years during fuel cycles 9 and 10. The licensee stated that the increase in neutron fluence during this period is more than offset by the fluence margins resulting from operating at a lower core power density over the lifetime of the unit. The NRC staff reviewed the margins and agrees with the licensee that operating with a lower core power density over the lifetime of the unit offsets the increase in fluence associated with an active fuel to upper core plate distance of less than 12.2 inches. Therefore, the NRC staff accepts the licensee's resolution to the Salem, Unit No. 1, exception.

The NRC staff finds that the licensee adequately demonstrated that Salem, Unit Nos. 1 and 2, are consistent with the MRP-227-A assumptions regarding core loading/core design, neutron fluence, and heat generation rates, with one exception for Salem, Unit 1, which the NRC staff finds acceptable.

As discussed above, the licensee has adequately demonstrated the applicability of the I&E Guidelines of MRP-227-A to the Salem, Unit Nos. 1 and 2, RVI components. Accordingly, the NRC staff determined that the licensee has adequately resolved applicant/licensee action item 1.

3.4.2 Assessment of Resolution to Applicant/Licensee Action Item 2

MRP-227-A, Section 4.2.2, "PWR Vessel Internal Components Within the Scope of License Renewal," states:

> As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR [license renewal] for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. **This is Applicant/Licensee Action Item 2**.

Salem, Unit Nos. 1 and 2, are Westinghouse-designed plants; therefore, only Table 4-4 of MRP-191 is applicable. The licensee stated in RVI AMPs Section 6.2.2, "SE Applicant/Licensee Action Item 2: PWR Vessel Internal Components within Scope of License Renewal," that a detailed tabulation of the RVI components was completed and compared to typical

Westinghouse PWR RVI components in Table 4-4 of MRP-191. The licensee identified that several Salem, Unit Nos. 1 and 2, RVI components, have the potential to be Grade CF8 CASS (CF8 CASS). The licensee provided these two components as examples: the CRGT cards and the brackets/clamps/terminal blocks/conduit straps (these are grouped into one unit in MRP-191 and MRP-227-A). Table 4-4 of MRP-191 identified these same components as Type 304 stainless steel.

The licensee stated that it formed an expert panel, consisting of knowledgeable Westinghouse individuals with expertise in the areas of reactor internals designs, materials age-related degradation mechanisms, safety analysis, and asset management, that reviewed the components having the potential to be CF8 CASS, and, based on the process in MRP-191, reached a 100 percent consensus that the MRP-227-A inspection category for these components does not change even if they are CF8 CASS. Based on this, the licensee concludes that there is no need for modification of the MRP-227-A program for these components.

The NRC staff agrees with the licensee's conclusion because it vetted the issue to an expert panel, consistent with the process in MRP-191, which concluded no change to the MRP-227-A program is needed. A discussion of Salem, Unit Nos. 1 and 2, RVI components being potentially CF8 CASS material is further addressed in the resolution of applicant/licensee action item 7 in Section 3.4.6 of this assessment. Based on the discussion above, the NRC staff determined that the licensee has adequately resolved applicant/licensee action item 2.

3.4.3 Assessment of Resolution to Applicant/Licensee Action Item 3

MRP-227-A, Section 4.2.3, "Evaluation of the Adequacy of Plant-Specific Existing Programs," states:

> As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analysis and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). **This issue is Applicant/Licensee Action Item 3**.

The licensee stated in Section 6.2.3, "SE Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs," of the Salem RVI AMPs that the Alloy X-750 CRGT support pins were replaced by support pins made of strain-hardened Type 316 stainless steel material. MRP-227-A, Table 3-3, "Final disposition of Westinghouse internals," identifies only Alloy X-750 CRGT support pins as requiring monitoring for aging during the PEO. Therefore, by letter dated March 31, 2015, the NRC staff requested the licensee to provide the plant-specific evaluation of the existing program under which the replacement Type 316 stainless steel CRGT support pins are currently inspected. Additionally, the NRC staff requested an evaluation of the adequacy of the existing inspection program to ensure that the aging degradation is adequately managed during the PEO for the Type 316 stainless steel CRGT support pins.

By letter dated May 28, 2015, the licensee responded by indicating that Subsection 4.4.3, "Westinghouse Components," of MRP-227-A states that guidance for monitoring the CRGT support pins is limited to plant-specific recommendations. Specifically, Subsection 4.4.3 of MRP-227-A states that subsequent performance monitoring of the support pins should follow the recommendations of the original equipment manufacturer (OEM). Additionally, the licensee responded that the OEM does not require subsequent inspection of the Type 316 stainless steel CRGT support pins; that long-term behavior of the Type 316 stainless steel CRGT support pins has been extensively studied; and that all potential degradation mechanisms, including stress corrosion cracking, wear, and fatigue have been assessed. The licensee, therefore, concluded that the Type 316 stainless steel CRGT support pins will perform their intended functions during the PEO without a need for post-installation inspections.

The NRC staff finds that the licensee followed the recommendation of the OEM, and therefore the guidance in Subsection 4.4.3 of MRP-227-A, for plant-specific performance monitoring of the Type 316 stainless steel CRGT support pins. Furthermore, Type 316 stainless steel CRGT support pins are in Category A in MRP-191 and binned into the "No Additional Measures" category in Figure 2-2 of MRP-227-A. Based on the discussion above, the NRC staff determined there is reasonable assurance that the licensee will adequately manage the aging of the Salem, Unit Nos. 1 and 2, CRGT support pins during the PEO. Accordingly, the NRC staff determined that the licensee has adequately resolved applicant/licensee action item 3.

3.4.4 Assessment of Resolution to Applicant/Licensee Action Item 4 and Applicant/Licensee Action Item 6

MRP-227-A applicant/licensee action item 4 and applicant/licensee action item 6are only applicable to plants designed by Babcock and Wilcox and, therefore, not applicable to Salem, Unit Nos. 1 and 2.

3.4.5 Assessment of Resolution to Applicant/Licensee Action Item 5

MRP-227-A, Section 4.2.5, "Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components," states:

> As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. **This is Applicant/Licensee Action Item 5**.

The licensee stated in Section 6.2.5, "SE Applicant/Licensee Action Item 5: Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI

Components," of the Salem RVI AMPs that Type 304 stainless steel hold down springs (HDS) are used at Salem, Unit Nos. 1 and 2, and that acceptance criteria for the physical measurements have been met and are consistent with the current licensing basis for Salem, Unit Nos. 1 and 2. The licensee referenced Westinghouse proprietary document LTR-RIDA-14-14, "Salem Units 1 and 2 – Lower Internals Hold-down Spring Height Criteria," Revision 1, which contains the acceptance criteria for the HDS measurements.

By letter dated March 31, 2015, the NRC staff requested the licensee to provide an explanation of the methodology for developing the acceptance criteria for the measurement of loss of compressibility of the Type 304 stainless steel HDS. By letter dated May 28, 2015, the licensee stated that the HDS height is assumed to decrease linearly over time and that the actual HDS height at the beginning of startup and the required HDS height at the end of PEO are considered when determining the minimum HDS height requirement before or during the PEO. PSEG stated that the methodology follows the general approach in Westinghouse TR WCAP-17096-NP, Revision 2 (Reference 22), in determining the acceptance criteria of the HDS. The approach in WCAP-17096-NP, Revision 2, requires plant-specific data that include direct, repeated measurements of spring height, historical information on spring height, and the required spring hold-down force. By letter dated May 3, 2016 (Reference 16), the NRC staff issued its SE of TR WCAP-17096-NP, Revision 2.

The NRC staff confirmed that the licensee's HDS measurement methodology is consistent with the approach in NRC-approved TR WCAP-17096-NP, Revision 2. Therefore, the NRC staff determined there is reasonable assurance that the HDS will perform their intended function of maintaining compression during the PEO. Accordingly, the NRC staff determined that the licensee has adequately resolved applicant/licensee action item 5.

3.4.6 Assessment of Resolution to Applicant/Licensee Action Item 7

MRP-227-A, Section 4.2.7, "Plant-Specific Evaluation of CASS Materials," states:

> As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W in-core monitoring instrumentation guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of

MRP-227. **This is Applicant/Licensee Action Item 7.**

PSEG evaluated the Salem, Unit Nos. 1 and 2, RVI components for potentially being made of CASS and their susceptibility to thermal embrittlement (TE) in the Salem RVI AMPs Section 6.2.7, "SE Applicant/Licensee Action Item 7: Plant-Specific Evaluation of Cast Austenitic Stainless Steel (CASS) Materials." As a result of the evaluation, the licensee determined that some RVI components are potentially made of CF8 CASS. The licensee listed these RVI components in Table 6-2, "Summary of SGS Unit 1 [Unit 2] CASS Components and their Susceptibility to TE," of the Salem RVI AMPs, respectively. Because the licensee could not locate certified material test reports needed to determine the susceptibility to TE of these potentially CF8 CASS RVI components, the licensee conservatively assumed they were susceptible to TE, and if fluence levels for the particular component are high, also to irradiation embrittlement (IE). Of these potentially CF8 CASS RVI components, three components listed below are of particular concern to the NRC staff for the reasons given. For clarity, the NRC staff presents a generic assessment of all potentially CF8 CASS RVI components in the next section, followed by a separate assessment of each of the three components of particular concern.

Component	Salem Unit No.	Reason for NRC Staff Concern	NRC Staff Assessment
CRGT Lower Flange	1 and 2	Binned in Category B in MRP-191	See Section 3.4.6.2 below
CRGT Cards	1 and 2	Binned in Category C in MRP-191	See Section 3.4.6.3 below
Lower Support Column Bodies	1 (73 columns)	Specifically called out for evaluation in application/licensee action item 7 of MRP-227-A	See Section 3.4.6.4 below

3.4.6.1 Assessment of Potential CASS Components Identified In Table 6-2

In its letter dated March 31, 2015, the NRC staff pointed to new staff guidance issued on June 11, 2014 (Reference 23), regarding evaluation of CASS materials due to the combined effects of TE and IE. The NRC staff requested the licensee to address the difference between the June 2014 guidance for TE susceptibility of CASS and the evaluation the licensee performed for Salem, Unit Nos. 1 and 2, CASS RVI components, which was based on guidance contained in an NRC letter dated May 19, 2000 (Reference 24, known as the "Grimes letter"). PSEG provided a response in its letters dated May 28, 2015; October 5, 2016; and May 3, 2017. The licensee leveraged the latest screening criteria for CASS RVI components (applicable to CF3 and CF8 grades) in the NRC staff's SE of BWRVIP-234 (Reference 28). According to this latest screening criteria, there is no significant loss of fracture toughness for statically cast CF3 CASS and CF8 CASS with less than 20 percent delta ferrite exposed to fluence levels between 0.00015 displacements per atom (dpa) to 1 dpa. The licensee then explained that since the CF8 CASS lower support column bodies (LSCs) would likely have delta ferrite contents below 20 percent, based on the conclusions of the statistical assessment of CASS RVI components in report PWROG-15032-NP (Reference 25) and the NRC staff assessment dated September 9, 2016 (Reference 26), the LSCs would not be considered susceptible to TE. While the licensee discussed only the LSCs, the NRC staff determined that the conclusions of PWROG-15032-NP apply to all the CASS RVI components the licensee identifed in Table 6-2 of its submittal. The NRC staff noted that the CASS RVI components in Table 6-2 of the submittal are all grade CF8, and, therefore, the updated screening criteria in the NRC staff's SE of BWRVIP-234 are applicable to them. The NRC staff reviewed the relevant sections in the NRC staff's SE of BWRVIP-234 and the NRC staff assessment of PWROG-15032-NP, and concluded that the CF8 CASS RVI components in Table 6-2 of the submittal are not susceptible to TE.

The licensee did not address loss of fracture toughness due to IE in its responses. However, the NRC staff noted that the CF8 CASS RVI components in Table 6-2 of the submittal do not screen in for IE in MRP-191 except the three listed below and those addressed in Sections 3.4.6.2, 3.4.6.3, and 3.4.6.4 of this assessment. The NRC staff verified that the three CF8 CASS RVI components below were determined to be susceptible to IE and categorized as Category A in MRP-191:

- Top Mounted Mixing Device, Mixing Device
- Upper Support Column – Orifice Base
- Upper Support Column – Mixing Bases

Category A components are further binned into the "No Additional Measures" category in Figure 2-2 of MRP-227-A. Therefore, the NRC staff determined that IE has already been addressed during the categorization process in MRP-227-A for these three CF8 CASS RVI components.

It is noted that the "Upper Support Column – Orifice Base," discussed in the August 11, 2014, submittal is equivalent to "Column Bases" in MRP-191.

3.4.6.2 Assessment of CRGT Lower Flange

The CRGT lower flange is binned into Category B in MRP-191, for which the significance of aging degradation is moderate. Therefore, the CRGT lower flange must be adequately managed during the PEO. The licensee included the welds in the CRGT lower flange (the welds being the lead indicators of degradation in the CRGT lower flange) as a component in the "Primary" inspection category in Table C-1 of Attachments 1 and 2 of the submittal, consistent with MRP-227-A. CRGT lower flanges made of CF8 CASS were considered in the MRP-191 categorization basis for MRP-227-A. The NRC staff, therefore, determined that CRGT lower flanges made of CF8 CASS are already in the proper MRP-227-A category. Additionally, the NRC staff determined in Section 3.4.6.1 of this assessment that the fracture toughness of CF8 CASS remains high enough throughout its service life that the structural integrity of components made of the CF8 CASS due to IE and TE will not likely be challenged. Accordingly, the NRC staff determined that the licensee will adequately manage the functionality of the Salem, Unit Nos. 1 and 2, CRGT lower flange during the PEO.

3.4.6.3 Assessment of CRGT Cards

The CRGT cards are binned into Category C in MRP-191, for which the significance of aging degradation is moderate to high. Therefore, the CRGT cards must be adequately managed during the PEO. Although the licensee included the CRGT cards as a component in the "Primary" inspection category in Table C-1 of the Salem RVI AMPs consistent with MRP-227-A, the only degradation mechanism considered was wear because in the MRP-191 categorization, only CRGT cards made of wrought stainless steel were considered. Since the CRGT cards are components the licensee identified as being potentially made of CF8 CASS and assumed susceptible to loss of fracture toughness due to TE, by letter dated July 7, 2016, the NRC staff requested the licensee to evaluate the susceptibility of CRGT cards to non-ductile cracking due to lower fracture toughness, including the effects of IE, and discuss how their functionality will be maintained during the PEO. Instead of a functionality analysis, the licensee, in its letter dated January 13, 2017, leveraged the statistical assessment of CASS RVI components in report PWROG-15032-NP and the NRC staff assessment dated September 9, 2016, to evaluate loss of fracture toughness of the CRGT cards. Among the statistical analyses in PWROG-15032-NP are calculations of saturated fracture toughness ($J_{2.5}$) based on a 95/95 confidence level for RVI components that are made of CASS. PWROG-15032-NP and the NRC staff assessment, citing NUREG/CR-7185 (Reference 27), state that a CF8 casting of unknown chemistry and ferrite levels of up to 40 percent would still have a saturated fracture toughness of $J_{2.5}$ = 329 kJ/m^2, which is above the TE screening level of $J_{2.5}$ = 255 kJ/m^2 specified in the Grimes letter. Therefore, the licensee stated that CRGT cards potentially made of CF8 CASS should not be considered susceptible to loss of fracture toughness due to TE. The NRC staff reviewed the licensee's responses and confirmed they are consistent with the relevant sections in PWROG-15032-NP and the NRC staff assessment. The NRC staff agrees with the licensee's assessment that the CRGT cards should not be considered susceptible to loss of fracture toughness due to TE.

Regarding loss of fracture toughness of the CRGT cards due to IE, the licensee used the updated screening criteria for loss of fracture toughness applicable to CF3 and CF8 CASS in the NRC staff's SE of BWRVIP-234, which considers the synergistic effect of TE and IE. According to this updated screening criteria, there is no significant loss of fracture toughness for statically cast CF3 CASS and CF8 CASS with less than 20 percent delta ferrite exposed to fluence levels between 0.00015 dpa to 1 dpa. The licensee stated that because the CRGT cards are significantly above the active core, they are exposed to fluence levels significantly below 1 dpa. The licensee stated that given the CRGT cards are below a 1 dpa fluence level, and given that CF8 CASS components are likely to have delta ferrite less than 20 percent, based on the statistical evaluations in PWROG-15032-NP, the CRGT cards screen out for significant loss of fracture toughness per the updated screening criteria. The NRC staff reviewed the relevant portions of PWROG-15032-NP, the NRC staff assessment of PWROG-15032-NP, and the NRC staff's SE of BWRVIP-234. The NRC staff accepts that the CRGT cards are exposed to fluence levels less than 1 dpa because they are above the active core, typical of Westinghouse reactor designs. Also, the NRC staff agrees that CF8 CASS components are likely to have delta ferrite less than 20 percent, based on the fact that only two of 404 statically cast CF8 heats have delta ferrite content above 20 percent in the NRC staff assessment of PWROG-15032-NP. The NRC staff assessment of PWROG-15032-NP states that the two heats that have delta ferrite content above 20 percent would still have significant fracture toughness throughout their service life. Therefore, the NRC staff concludes that it is reasonable to screen out the CRGT cards for significant loss of fracture toughness due to TE and IE per the updated screening criteria in the NRC staff's SE of BWRVIP-234.

Based on the above discussion that the Salem, Unit Nos. 1 and 2, CRGT cards are not susceptible to loss of fracture toughness due to TE or IE (or their synergistic effect), a functionality analysis for these components is not needed, and the MRP-191 categorization for CRGT cards is unaffected. Therefore, the NRC staff concludes that the licensee has demonstrated reasonable assurance that the Salem, Unit Nos. 1 and 2, CRGT cards will be adequately managed during the PEO.

3.4.6.4 Assessment of Lower Support Column Bodies

The licensee discussed in Section 6.2.7 of the Salem RVI AMPs that the certified material test reports cannot be located for 73 of the 96 LSCs of Salem, Unit No. 1. The certified material test reports contain composition data that would determine whether the 73 LSCs, assumed CF8

CASS, are susceptible to loss of fracture toughness due to TE. Since the certified material test reports cannot be located for the 73 LSCs, the licensee assumed they were susceptible to loss of fracture toughness due to TE. Therefore, by letter dated March 31, 2015, the NRC staff requested the licensee to explain how loss of fracture toughness due to TE (and also due to IE per applicant/licensee action item 7 of the SE in MRP-227-A) of the LSCs will be managed during the PEO.

In an effort to generically address the functionality of CASS LSCs due to TE and IE, the PWROG prepared proprietary report PWROG-14048-P, "Functionality Analysis: Lower Support Columns," and by letter dated March 13, 2015, submitted it to the NRC for information only (Reference 29). In a summary assessment of the report dated December 17, 2015 (Reference 30), the NRC staff determined that the generic flaw tolerance analysis in PWROG-14048-P, Revision 0, utilized conservative assumptions to demonstrate that the likelihood of failure of LSCs is low during the PEO. The NRC staff requested the licensee to demonstrate how the generic flaw tolerance analysis in PWROG-14048-P is applicable to the 73 Salem, Unit No. 1, CASS LSCs that are assumed to be susceptible to TE using plant-specific parameters (such as LSC geometry and number of LSCs) and conditions (such as loading conditions and LSC stresses).

By letter dated January 13, 2017, PSEG committed to submit a copy of the technical report associated with the revision to report PWROG-14048-P, Revision 0, to the NRC 60 days following the licensee's receipt of the revision to PWROG-14048-P. The licensee stated in the commitment that the technical report would contain Salem, Unit Nos. 1 and 2, applicable flaw tolerance analysis consistent with the revision to PWROG-14048-P, and the licensee would respond to the NRC staffs request that Salem-specific LSC parameters and loading conditions be used or considered in the analysis.

By letter dated March 1, 2017, the PWROG submitted PWROG-14048-P, Revision 1, for information only (Reference 31). The licensee referenced the submittal of PWROG-14048-P, Revision 1, in a letter dated May 3, 2017, indicating that this completed its commitment made in the January 13, 2017, letter. The licensee stated in the May 3, 2017, letter that PWROG-14048-P, Revision 1, contains evaluations of LSCs that bound or are consistent with the geometric and loading conditions of the Salem, Unit Nos. 1 and 2, LSCs. Furthermore, the licensee stated that the flaw tolerance evaluations in Revision 1 of PWROG-14048-P demonstrate low likelihood of failure of the Salem, Unit Nos. 1 and 2, LSCs even with the effects of TE and IE considered, and that the LSC redundancy analysis in Revision 1 of PWROG-14048-P demonstrates that sufficient redundancy exists in the lower support structure (which includes the LSCs), should some LSCs fail.

The NRC staff issued a staff assessment dated September 8, 2017 (Reference 32), of PWROG-14048-P, Revision 1. In this summary assessment, the NRC staff made the same conclusions about LSC functionality as it did in its staff assessment of the report dated December 17, 2015, for PWROG-14048-P, Revision 0, but extended to all Westinghouse and CE LSC designs of participating members of the PWROG. However, one of the conclusions in the NRC staff assessment of PWROG-14048-P, Revision 1, is that the redundancy analysis in the report did not address the effect of the bending moment in LSC buckling. In the section titled "Assessment of Change 4" in the NRC staff assessment of PWROG-14048-P, Revision 1, the NRC staff noted the high bending stresses for the faulted conditions, which could lead to failure of the LSCs due to compressive yielding, but further observed that the faulted condition analyzed in the report is very conservative and an unlikely condition since loss-of-coolant accident and seismic events are assumed to occur at the same time. Furthermore, the NRC staff determined that the flaw tolerance evaluation in the report demonstrated that the likelihood of full-section failure of the LSCs is low. This means that the likelihood of having an LSC configuration with broken LSCs is low. Since high bending stresses occur under faulted loads for cases with broken LSCs, the likelihood of having LSCs subject to high bending stresses is low. Therefore, the NRC staff determined that the LSCs will be adequately managed during the PEO.

The NRC staff verified that Salem, Unit Nos. 1 and 2, are active participants in the PWROG program to address the functionality analysis of LSCs. Therefore, the NRC staff determined that the LSC design of Salem, Unit No. 1 and 2, was considered in the bounding LSC functionality analyses in PWROG-14048-P, Revision 1. Accordingly, the NRC staff determined that the licensee has successfully fulfilled the commitment it made in the January 13, 2017, letter and that the LSCs will be adequately managed during the PEO.

3.4.6.5 Conclusion of Section 3.4.6

Based on the discussions in Sections 3.4.6.1 through 3.4.6.4 of this assessment, the NRC staff determined that the licensee will adequately manage the functionality of Salem, Unit Nos. 1 and 2, CASS RVI components during the PEO. Accordingly, the NRC staff determined that the licensee has adequately resolved applicant/licensee action item 7.

3.4.7 <u>Assessment of Resolution to Applicant/Licensee Action Item 8</u>

MRP-227-A, Section 4.2.8, "Submittal of Information for Staff Review and Approval," states:

> As addressed in Section 3.5.1 of this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. **This is Applicant/Licensee Action Item 8**.

The licensee stated in Section 6.2.8, "SE Applicant/Licensee Action Item 8: Submittal of Information for Staff Review and Approval," of the Salem, Unit Nos. 1 and 2, RVI AMPs that the Salem units fall under Category B per NRC Regulatory Issue Summary (RIS) 2011-07 (Reference 33). RIS 2011-07 reminds license renewal applicants that Category B plants must submit RVI AMPS based on the I&E Guidelines in MRP-227-A. Regarding the information in MRP-227-A, Section 3.5.1, as stated in applicant/licensee action item 8 above, the NRC staff confirmed that the Salem, Unit Nos. 1 and 2, RVI AMPs address the ten program elements of the GALL Report. Accordingly, the NRC staff determined that the licensee has adequately resolved applicant/licensee action item 8.

4.0 <u>CONCLUSION</u>

As described above, the NRC staff has reviewed the AMPs for Salem, Unit Nos. 1 and 2, RVI components and concludes that the Salem, Unit Nos 1 and 2, RVI AMPs are acceptable because they are consistent with the I&E Guidelines of MRP-227-A. The NRC staff finds that the licensee has adequately addressed and resolved the eight applicant/licensee action items specified in MRP-227-A.

The NRC staff's approval of the Salem, Unit Nos. 1 and 2, RVI AMPs does not reduce, alter, or otherwise affect current ASME Code, Section XI, ISI requirements, or any Salem, Unit Nos. 1 and 2, specific licensing basis requirements related to ISI.

5.0 REFERENCES

1. Letter from John F. Perry, PSEG, to NRC, "Submittal of PWR Vessel Internals Inspection Plans for Aging Management of Reactor Internals at Salem Generating Station, Units 1 and 2," August 11, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14224A667).

2. Letter from John F. Perry, PSEG, to NRC, "Response to Salem Nuclear Generating Station, Unit Nos. 1 and 2 – Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals," May 28, 2015 (ADAMS Accession No. ML15148A426).

3. Letter from John F. Perry, PSEG, to NRC, "Response to Request for Additional Information, RAI-4 Re: Aging Management Program Plan for Reactor Vessel Internals, March 31, 2015," March 23, 2016 (ADAMS Accession No. ML16083A194).

4. Letter from Eric Carr, PSEG, to NRC, "Response to Request for Additional Information, RAI-8 – RAI-11, Re: Aging Management Program Plan for Reactor Vessel Internals," October 5, 2016 (ADAMS Accession No. ML16279A092).

5. Letter from Paul J. Davison, PSEG, to NRC, "Supplemental Information for Response to Request for Additional Information, RAI-8 – RAI-11, Re: Aging Management Program Plan for Reactor Vessel Internals," January 13, 2017 (ADAMS Accession No. ML17013A251).

6. Letter from Paul J. Davison, PSEG, to NRC, "Supplemental Information for Response to Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals," May 3, 2017 (ADAMS Accession No. ML17123A070).

7. EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), 1022863, Final Report, December 2011 (ADAMS Package Accession No. ML120170453).

8. NRC NUREG-2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station," June 2011 (ADAMS Accession No. ML11166A135).

9. EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227), Revision 0, January 12, 2009 (ADAMS Package Accession No. ML090160212).
10. NRC Revision 1 to Final Safety Evaluation of EPRI Report MRP-227, Revision 0, December 16, 2011 (ADAMS Accession No. ML11308A770).

11. Letter from NRC to Thomas Joyce, PSEG, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 – Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals," March 31, 2015 (ADAMS Accession No. ML15069A181).

12. Letter from NRC to Peter P. Sena, III, PSEG, "Salem Nuclear Generating Station, Unit Nos. 1 and 2 – Request for Additional Information Re: Aging Management Program Plan for Reactor Vessel Internals," July 7, 2016 (ADAMS Accession No. ML16188A415).

13. NRC NUREG-1801, "Generic Aging Lessons Learned (GALL) Report – Final Report," Revision 2, December 2010 (ADAMS Accession No. ML103490041).

14. EPRI Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191), 1013234, Technical Report," November 2006 (ADAMS Accession No. ML091910130).

15. Westinghouse Technical Bulletin, TB-14-5, Revision 0, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," August 25, 2014 (ADAMS Accession No. ML15222A885) (non-public).

16. Letter from NRC to Anne Demma, EPRI, "Final Safety Evaluation of WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements,'" May 3, 2016 (ADAMS Accession No. ML16061A243).

17. EPRI Letter MRP 2016-022, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt inspections for Tier 1 plants as Defined in Westinghouse NSAL 16-01," July 27, 2016 (ADAMS Accession No. ML16211A054).

18. EPRI Letter MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013 (ADAMS Accession No. ML13322A454).

19. Office of Nuclear Reactor Regulation Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering [(CE)] and Westinghouse [Electric Company (Westinghouse)] Pressurized Water Reactor Designs," and MRP-227-A, "Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs," November 7, 2014 (ADAMS Accession No. ML14309A484).

20. Pressurized Water Reactor Owners Group Report PWROG-15105-NP, Revision 0, "PA-MSC-1288, PWR RV Internals Cold-Work Assessment, Materials Committee, PA-MSC-1288," April 2016 (ADAMS Accession No. ML16222A300).

21. Office of Nuclear Reactor Regulation Summary Assessment of Pressurized-Water Reactor Owners Group - 15105-NP, Revision 0, "PA-MSC-1288 Pressurized-Water Reactor Vessel Internals Cold Work Assessment," Staff Assessment," April 21, 2017 (ADAMS Accession No. ML17081A010).

22. Westinghouse Report WCAP-17096-NP-A, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August 2016 (ADAMS Accession No. ML16279A320).

23. NRC document, "NRC position on Aging Management of CASS Reactor Vessel Internal Components," June 11, 2014 (ADAMS Accession No. ML14163A112).

24. Letter from NRC to Douglas J. Walters, Nuclear Energy Institute, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,'" May 19, 2000 (ADAMS Accession No. ML003717179).

25. PWROG-15032-NP, Revision 0, "PA-MSC-1288, Statistical Assessment of PWR RV Internals CASS Materials," Parts 1 and 2, November 2015 (ADAMS Accession

Nos. ML16068A245 and ML16068A246).

26. Office of Nuclear Reactor Regulation Staff Assessment of the Pressurized Water Reactor Owner's Group Report PWROG-15032-NP, Revision 0, "PA-MSC-1288, Statistical Assessment of PWR RV Internals CASS Materials," September 9, 2016 (ADAMS Accession No. ML16250A001).

27. NUREG/CR-7185, "Effects of Thermal Aging and Neutron Irradiation on Crack Growth Rate and Fracture Toughness of Cast Stainless Steels and Austenitic Stainless Steel Welds," July 2015 (ADAMS Accession No. ML15202A007).

28. Letter from NRC to Tim Hanley, Chairman, BWR Vessel and Internals Project, "Final Safety Evaluation of the BWRVIP-234: Thermal Aging and Neutron Embrittlement Evaluation of Cast Austenitic Stainless Steel for BWR Internals," June 22, 2016 (ADAMS Accession No. ML16096A002).

29. Letter from Jack Stringfellow, PWROG, to NRC, "PWR Owners Group: Submittal of Report PWROG-14048-P, Revision 0, 'Functionality Analysis: Lower Support Columns' to the NRC for Information Only (PA-MSC-1103)," March 13, 2015 (ADAMS Accession No. ML15077A113).

30. Memorandum from Office of Nuclear Reactor Regulation, "Summary Assessment of PWROG-14048-P, Revision 0, Functionality Analysis: Lower Support Columns," December 17, 2015 (ADAMS Accession No. ML15334A462).

31. Letter from Jack Stringfellow, PWROG, to NRC, "PWR Owners Group: Submittal of PWROG-14048-P, Revision 1, 'Functionality Analysis: Lower Support Columns' to the NRC for Information Only (PA-MSC-1103)," March 1, 2017 (ADAMS Accession No. ML17066A266).

32. NRC Staff Assessment of PWROG-14048-P, Revision 1, "Functionality Analysis: Lower Support Columns," September 8, 2017 (ADAMS Accession No. ML17251A905).

33. NRC Regulatory Issue Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 21, 2011 (ADAMS Accession No. ML111990086).

Principal Contributor: David Dijamco

Date: November 21, 2017

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 – STAFF ASSESSMENT OF THE REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM (CAC NOS. MF5149 AND MF5150; EPID L-2014-LRL-0001) DATED NOVEMBER 21, 2017

DISTRIBUTION:
Public
RidsNrrDssStsb Resource
RidsRgn1MailCenter Resource
RidsNrrPMSalem Resource
DDijamco, NRR
RidsNrrDmlrMvib Resource
RidsACRS_MailCTR Resource
RidsNrrDorlLpl1 Resource
RidsNrrDmlrMvib Resource
RidsNrrLALRonewicz Resource
MYoo, NRR
TBuchanan, NRR

ADAMS Accession No.: ML17320A859 *by memorandum dated

OFFICE	DORL/LPL1/PM	DORL/LPL1/LA	DE/EVIB/BC(A)*	DLR/RARB/BC(A)*	DORL/LPL1/BC
NAME	CParker	LRonewicz	SRuffin	TBuchanan	JDanna
DATE	11/20/2017	11/16/2017	09/28/2017	09/28/2017	11/21/2017

OFFICIAL RECORD COPY